For the Nuclear Regulatory Commission.

Dennis M. Crutchfield,

Director, Division of Reactor Program Management, Office of Nuclear Reactor Regulation.

[FR Doc. 95–26421 Filed 10–24–95; 8:45 am]

BILLING CODE 7590–01–P

Biweekly Notice; Applications and Amendments to Facility Operating Licenses Involving No Significant Hazards Considerations

I. Background

Pursuant to Public Law 97–415, the U.S. Nuclear Regulatory Commission (the Commission or NRC staff) is publishing this regular biweekly notice. Public Law 97–415 revised section 189 of the Atomic Energy Act of 1954, as amended (the Act), to require the Commission to publish notice of any amendments issued, or proposed to be issued, under a new provision of section 189 of the Act. This provision grants the Commission the authority to issue and make immediately effective any amendment to an operating license upon a determination by the Commission that such amendment involves no significant hazards consideration, notwithstanding the pendency before the Commission of a request for a hearing from any person.

This biweekly notice includes all notices of amendments issued, or proposed to be issued from September 29, 1995, through October 13, 1995. The last biweekly notice was published on October 11, 1995 (60 FR 52927).

Notice of Consideration of Issuance of Amendments to Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing

The Commission has made a proposed determination that the following amendment requests involve no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. The basis for this proposed determination for each amendment request is shown below.

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination.

Normally, the Commission will not issue the amendment until the expiration of the 30-day notice period. However, should circumstances change during the notice period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the Commission may issue the license amendment before the expiration of the 30-day notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The final determination will consider all public and State comments received before action is taken. Should the Commission take this action, it will publish in the Federal Register a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to take this action will occur very infrequently.

Written comments may be submitted by mail to the Rules Review and Directives Branch, Division of Freedom of Information and Publications Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and should cite the publication date and page number of this Federal Register notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the NRC Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By November 24, 1995, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

As required by 10 CFR 2.714, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following factors: (1) The nature of the petitioner's right under the Act to be made a party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to 15 days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

Not later than 15 days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner shall provide a brief explanation of the bases of the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner intends to rely to establish those facts or expert opinion. Petitioner must provide sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses.

If a hearing is requested, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held.

If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment.

If the final determination is that the amendment request involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington DC, by the above date. Where petitions are filed during the last 10 days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1–(800) 248–5100 (in Missouri 1–(800) 342–6700). The Western Union operator should be given Datagram Identification Number N1023 and the following message addressed to (*Project Director*): petitioner's name and telephone number, date petition was mailed, plant name, and publication date and page number of this Federal Register notice. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to the attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for a hearing will not be entertained absent a determination by the Commission, the presiding officer or the Atomic Safety and Licensing Board that the petition and/or request should be granted based upon a balancing of factors specified in 10 CFR 2.714(a)(1)(i)–(v) and 2.714(d).

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room for the particular facility involved.

Arizona Public Service Company, et al., Docket Nos. STN 50–528, STN 50–529, and STN 50–530, Palo Verde Nuclear Generating Station, Units Nos. 1, 2, and 3, Maricopa County, Arizona

Date of amendments request: June 13, 1995, as supplemented by letter dated August 16, 1995.

Description of amendments request: The proposed amendments would extend allowed outage times (AOTs) for a safety injection tank (SIT), a low-pressure safety injection (LPSI) subtrain, and an emergency diesel generator (EDG) and add the bases for the extended AOTs.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The Safety Injection Tanks (SITs) are passive components in the Emergency Core Cooling System. The SITs are not an accident initiator in any accident previously evaluated. Therefore, this change does not involve a significant increase in the probability of an accident previously evaluated.

SITs were designed to mitigate the consequences of Loss of Coolant Accidents (LOCA). These proposed changes do not affect any of the assumptions used in deterministic LOCA analysis. Hence the consequences of accidents previously evaluated do not significantly increase.

The allowed outage time (AOT) extension for boron concentration outside the prescribed limits does not involve a significant increase in the consequences of an accident as evaluated and approved by the NRC in NUREG–1432, ''Standard Technical Specifications for Combustion Engineering Plants.'' These changes are applicable to PVNGS.

The changes pertaining to SIT inoperability based solely on instrumentation malfunction do not involve a significant increase in the consequences of an accident as evaluated and endorsed by the NRC in NUREG–1366, ''Improvements to Technical Specifications Surveillance Requirements,'' and Generic Letter 93–05, ''Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operations.'' These changes are applicable to PVNGS.

The AOT extension from one hour to 24 hours for a SIT that is inoperable due to reasons other than boron concentration not within limits or the inability to verify level or pressure does not involve a significant increase in the consequences of an accident. In order to fully evaluate the affect of the SIT AOT extension, probabilistic safety analysis (PSA) methods were utilized. The results of these analyses show no significant increase in the core damage frequencies (CDF). As a result, there would be no significant increase in the consequences of an accident previously evaluated. These analyses are detailed in CE NPSD–994, Combustion Engineering Owners Group ''Joint Applications Report for Safety Injection Tank AOT/STI Extension,'' May 1995.

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

This proposed change does not change the design, configuration, or method of operation of the plant. Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. The proposed change does not involve a significant reduction in a margin of safety.

The proposed changes do not involve a significant reduction in a margin of safety.

The proposed changes do not affect the limiting conditions for operation or their bases that are used in the deterministic analyses to establish the margin of safety. PSA evaluations were used to evaluate these changes. These evaluations demonstrated that the changes are either risk neutral or risk beneficial. These evaluations are detailed in CE NPSD–994.

The NRC staff has reviewed the licensee's analysis and, based on that review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendments request involve no significant hazards consideration.

Local Public Document Room location: Phoenix Public Library, 1221 N. Central Avenue, Phoenix, Arizona 85004.

Attorney for licensee: Nancy C. Loftin, Esq., Corporate Secretary and Counsel, Arizona Public Service Company, P.O. Box 53999, Mail Station 9068, Phoenix, Arizona 85072–3999.

NRC Project Director: William H. Bateman.

Carolina Power & Light Company, Docket No. 50–261, H. B. Robinson Steam Electric Plant, Unit No. 2, Darlington County, South Carolina

Date of amendment request: September 11, 1995.

Description of amendment request: The proposed change is to (1) modify a limiting condition for operation (LCO), TS Section 3.10.1.3, to provide for temporary conditions in which the full length control rod insertion limits (RILs) are exceeded due to automatic plant responses or conservative operator actions and (2) add an allowance for RILs to be exceeded for a time no greater than the time criteria established by the axial power distribution methodology or 1 hour, whichever is sooner. An action is added for the reactor to be placed in the hot shutdown condition within 6 hours if compliance with the RILs cannot be restored within the specified time period.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

This proposed change does not involve a significant hazards consideration for the following reasons.

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed change does not involve the addition or modification of plant equipment, nor does it alter the design, material, or operation of plant systems. No analyzed accidents are initiated by an entire control rod bank exceeding the RILs, due to automatic plant responses or conservative operator actions. The overall performance of the Reactor Control System, Power Distribution Control procedures, and Control Rod Drive System is not degraded. There is no increase in fatigue or number of operational cycles of equipment, and there is no change in system interfaces. The consequences of previously evaluated accidents are not increased since exceeding the RILs for a limited period is acceptable as the probability of a simultaneous occurrence of an independent accident is low. Therefore, an allowance for RILs to be exceeded for a maximum of one (1) hour does not affect the probability of occurrence or consequences of an analyzed accident.

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change adds an allowance for RILs to be exceeded for a maximum of one (1) hour. The proposed change does not involve the addition or modification of plant equipment, nor does it alter the design or operation of plant systems. The only procedural changes required will be those associated with recovery from the infrequent condition of exceeding the RILs. No new accident scenarios are introduced when the RILs are exceeded for a short period of time due to automatic plant responses or conservative operator actions because the probability of a simultaneous occurrence of an independent accident is low. Therefore, an allowance for RILs to be exceeded for a maximum of one (1) hour does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change does not involve a significant reduction in the margin of safety. The proposed change adds an allowance for RILs to be exceeded for a maximum of one (1) hour. The proposed change does not involve the addition or modification of plant equipment, nor does it alter the design or operation of plant systems. The overall performance of the Reactor Control System, Power Distribution Control, and Control Rod Drive System is not degraded. There is no increase in fatigue or number of operational cycles of equipment, and there is no change in system interfaces. When the RILs are exceeded for a limited time period, due to automatic plant responses or conservative operator actions, the margin of safety is not reduced because the probability of a simultaneous occurrence of an independent accident is acceptably low. Therefore, an allowance for RILs to be exceeded for a maximum of one (1) hour does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Hartsville Memorial Library, 147 West College Avenue, Hartsville, South Carolina 29550.

Attorney for licensee: R.E. Jones, General Counsel, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602.

NRC Project Director: David B. Matthews.

Commonwealth Edison Company, Docket Nos. STN 50–454 and STN 50–455, Byron Station, Unit Nos. 1 and 2, Ogle County, Illinois

Docket Nos. STN 50–456 and STN 50–457, Braidwood Station, Unit Nos. 1 and 2, Will County, Illinois

Date of amendment request: September 14, 1995.

Description of amendment request: The proposed amendment would allow the use of an alternate zirconium based fuel cladding, ZIRLO, and permit limited substitution of ZIRLO filler rods for fuel rods. The proposed amendment also includes a clarification and an editorial change.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

The methodologies used in the accident analyses remain unchanged. The proposed changes do not change or alter the design assumptions for the systems or components used to mitigate the consequences of an accident. Use of ZIRLO fuel cladding does not adversely affect fuel performance or impact nuclear design methodology. Therefore, accident analysis results are not impacted.

The operating limits will not be changed and the analysis methods to demonstrate operation within the limits will remain in accordance with NRC-approved methodologies. Other than the changes to the fuel assemblies, there are no physical changes to the plant associated with this Technical Specification change. A safety analysis will continue to be performed for each cycle to demonstrate compliance with all fuel safety design bases.

VANTAGE 5 fuel assemblies with ZIRLO clad fuel rods meet the same fuel assembly and fuel rod design bases as other VANTAGE 5 fuel assemblies. In addition, the 10 CFR 50.46 criteria are applied to the ZIRLO clad fuel rods. The use of these fuel assemblies will not result in a change to the reload design and safety analysis limits. Since the original design criteria are met, the ZIRLO clad fuel rods will not be an initiator for any new accident. The clad material is similar in chemical composition and has similar physical and mechanical properties as Zircaloy-4. Thus, the cladding integrity is maintained and the structural integrity of the fuel assembly is not affected. ZIRLO cladding improves corrosion performance and dimensional stability. No concerns have been identified with respect to the use of an assembly containing a combination of Zircaloy-4 and ZIRLO clad fuel rods. Since the dose predictions in the safety analyses are not sensitive to the fuel rod cladding material used, the radiological consequences of accidents previously evaluated in the safety analysis remain valid.

Replacing the reference to the Final Safety Analysis Report (FSAR) with a reference to the Updated Final Safety Analysis Report (UFSAR) is an editorial change to reflect the current document. Adding that reload fuel shall be similar in physical design to the initial core loading or previous cycle loading is a clarification. A reload analysis is completed for each cycle, in accordance with USNRC-approved methodologies.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

VANTAGE 5 fuel assemblies with ZIRLO clad fuel rods satisfy the same design bases as those used for other VANTAGE 5 fuel assemblies. All design and performance criteria continue to be met and no new failure mechanisms have been identified. The ZIRLO cladding material offers improved corrosion resistance and structural integrity.

The proposed changes do not affect the design or operation of any system or component in the plant. The safety functions of the related structures, systems, or components are not changed in any manner, nor is the reliability of any structure, system, or component reduced. The changes do not affect the manner by which the facility is operated and do not change any facility design feature, structure, or system. No new or different type of equipment will be installed. Since there is no change to the facility or operating procedures, and the safety functions and reliability of structures, systems, or components are not affected, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed changes do not involve a significant reduction in a margin of safety.

The use of Zircaloy–4, ZIRLO, or stainless steel filler rods in fuel assemblies will not involve a significant reduction in the margin of safety because analyses using NRC-approved methodology will be performed for each configuration to demonstrate continued operation within the limits that assure acceptable plant response to accidents and transients. These analyses will be performed using NRC-approved methods that have been approved for application to the fuel configuration.

Use of ZIRLO cladding material does not change the VANTAGE 5 reload design and safety analysis limits. The use of these fuel assemblies will take into consideration the normal core operating conditions allowed in the Technical Specifications. For each cycle reload core, the fuel assemblies will be evaluated using NRC-approved reload design methods, including consideration of the core physics analysis peaking factors and core average linear heat rate effects.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the requested amendments involve no significant hazards consideration. *Local Public Document Room location:* For Byron, the Byron Public Library District, 109 N. Franklin, P.O. Box 434, Byron, Illinois 61010; for Braidwood, the Wilmington Public Library, 201 S. Kankakee Street, Wilmington, Illinois 60481

Attorney for licensee: Michael I. Miller, Esquire; Sidley and Austin, One First National Plaza, Chicago, Illinois 60603.

NRC Project Director: Robert A. Capra.

Commonwealth Edison Company, Docket Nos. 50–373 and 50–374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois

Date of amendment request: August 28, 1995.

Description of amendment request: The proposed amendments would support elimination of the Main Steam Isolation Valve Leakage Control System (MSIV LCS) and instead use the main steamline drains and condenser to process MSIV leakage. The proposed changes would also increase the allowable MSIV leakage from 100 standard cubic feet per hour (scfh) for all four main steam lines to 100 scfh per steam line (400 scfh for all four main steam lines).

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated because:

The proposed changes involve eliminating the requirement for the Main Steam Isolation Valve Leakage Control System (MSIV LCS). This system is manually initiated following a design basis Loss of Coolant Accident (LOCA). Since operation of the LCS is initiated after the accident has already begun, elimination of that system will not affect the probability of a LOCA. The LCS only interfaces with the main steamlines, with the exception of one MSIV LCS power supply which supplies power to the Reactor Protection System Scram Discharge Volume high level scram. This power supply will remain in place after the MSIV LCS is isolated from the main steamlines. Therefore, since the only significant system interface is with the main steamlines, and the system does not impact the reliability of any plant equipment, elimination of that system will not cause an increase in the likelihood that any accident might occur.

The proposed change to increase the allowable MSIV leakage limit from 100 scfh through all four main steam lines to 100 scfh per main steam line (400 scfh total) will not increase the probability of an accident. MSIV operability will not be degraded with the allowed increased leakage.

The consequences of a LOCA are not significantly increased and do not exceed the previously accepted licensing criteria for this accident. General Electric has calculated the revised LOCA doses, which have been added to the previous LOCA doses. These resulting values are well below the acceptance criteria of 10CFR100 and 10CFR50, Appendix A.

The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated because:

The proposed changes require the use of the main steam piping and condenser to process MSIV leakage. The analyses presented provide assurance that this additional function does not compromise the reliability of those systems. They will therefore continue to function as intended and not be subject to an increased failure rate or a failure of a different kind than previously considered.

In addition, MSIV functionality will not be adversely impacted as a result of the increased leakage limit. The MSIVs are not being modified in any way and will continue to provide their intended isolation function.

The MSIV LCS will be cut and capped, which will completely isolate it from other plant systems. Future degradation of its associated piping would not impact any other system or create a failure not previously analyzed. However, piping seismic Class II over I criteria must be maintained for the abandoned MSIV LCS piping until it is removed from the plant.

The proposed changes do not involve a significant reduction in a margin of safety because:

The proposed change has been evaluated with respect to dose limits contained in 10CFR100 and 10CFR50, Appendix A. The revised dose calculations verify that the use of the main steam lines and the condenser for leakage control, in place of the MSIV LCS, and with an allowable total leakage of 400 scfh, maintains adequate margins to the criteria listed above.

Even though there is a reduction in the margin to safety, the new doses remain well within the criteria of 10 CFR 100 and 10 CFR 50, Appendix A. This reduction in margin is not significant when compared to the increased reliability and capability of the main steam lines and condenser as a method of treating MSIV leakage. The new leakage pathway is consistent with the philosophy of protection by multiple barriers for limiting fission product release to the environment. In addition, the new method is passive and does not require any new logic control or interlocks. The new pathway is also capable of handling a larger amount of leakage than the MSIV LCS, which was previously subject to concerns that it would not function at leakage rates higher than its design capacity, or at reactor pressures greater than 35 psig.

The revised calculated LOCA doses remain well within the regulatory limits for MSIV leakage rates of 400 scfh for all four main steam lines (100 scfh per steam line), and the margin to safety is not significantly reduced as a result of the proposed changes.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the requested amendments involve no significant hazards consideration.

Local Public Document Room location: Jacobs Memorial Library, Illinois Valley Community College, Oglesby, Illinois 61348.

Attorney for licensee: Michael I. Miller, Esquire; Sidley and Austin, One First National Plaza, Chicago, Illinois 60603.

NRC Project Director: Robert A. Capra.

Duke Power Company, Docket Nos. 50–369 and 50–370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina

Date of amendment request: September 1, 1995.

Description of amendment request: Generic Letter 88–16 provided guidance on removing cycle-specific parameters which are calculated using NRC-approved methodologies from the Technical Specifications (TS). The parameters are replaced in the TS with a reference to a named report which contains the parameters, and a requirement that the parameters remain within the limits specified in the report. The proposed changes incorporate NRC-approved methodologies, approved revisions to previously approved methodologies, or republished versions of previously approved methodologies into section 6.9.2 of the Oconee TS. The limits to which these methodologies are applied are (1) Axial Power Imbalance Protective Limits and Variable Low RCS Pressure Protective Limits, (2) Reactor Protective System Trip Setting Limits for the Flux/Flow/Imbalance and Variable Low Reactor Coolant System Pressure Trip Functions, and (3) Power Imbalance Limits. Since the proposed changes only incorporate NRC-approved methodologies into the TS, the licensee proposed that the changes are administrative in nature and can be assumed to have no impact, or potential impact, on the health and safety of the public.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

The proposed changes will not create a significant hazards consideration, as defined by 10 CRF 50.92, because:

(1) The proposed changes will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes are administrative in nature, and do not affect any system, procedure, or manipulation of any equipment which could affect the probability or consequences of any accident.

(2) The proposed changes will not create the possibility of any new or different kind of accident from any accident previously evaluated.

The proposed changes are administrative in nature, and cannot introduce any new failure mode or transient which could create any accident.

(3) The proposed changes will not involve a significant reduction in a margin of safety.

The proposed changes are administrative in nature, and will not affect any operating parameters or limits which could result in a reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Atkins Library, University of North Carolina, Charlotte (UNCC Station), North Carolina 28223.

Attorney for licensee: Mr. Albert Carr, Duke Power Company, 422 South Church Street, Charlotte, North Carolina 28242.

NRC Project Director: Herbert N. Berkow.

Entergy Operations, Inc., Docket No. 50–368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas

Date of amendment request: September 25, 1995.

Description of amendment request: The proposed amendment adds a repair limit for circumferential cracks in steam generator tubes. It deletes the requirement to repair cracks that are within the repair limit. The proposed amendment also reduces the primary-to-secondary leak rate limit.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

Criterion 1—Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

Consistent with draft Regulatory Guide (RG) 1.121, ''Basis for Plugging Degraded PWR Steam Generator Tubes,'' the traditional maximum depth based criteria for steam generator tube repair implicitly ensures that tubes accepted for continued service will retain adequate structural and leakage integrity during normal operating, transient, and postulated accident conditions. It is recognized that defects in tubes permitted to remain in service occasionally grow through-wall and develop small leaks. Limits on allowable primary-to-secondary leakage established in the technical specifications ensure timely plant shutdown before the structural and leakage integrity of the affected tube is challenged.

The proposed change to implement a circumferential crack repair limit in the expansion transition region for ANO–2 meets the criteria of RG 1.121. The 40% degraded area repair limit was determined by performing a structural analysis per the recommendations of the RG and applying the following uncertainties: 95% lower bound material properties, 95% lower bound burst curve, 95% lower bound eddy current measurement uncertainties, and 95% upper bound crack growth rate. The analysis demonstrates that tube leakage and conditional probability of burst are acceptably low during either normal operation or the most limiting accident condition, a postulated main steam line break (MSLB) event.

As part of the implementation of the circumferential crack repair limit, the distribution of End-of-Cycle (EOC) circumferential indications in the expansion transition region will be used to calculate the primary-to-secondary leakage. The allowable leakage is bounded by the maximum leakage which results in doses within the applicable dose limits (10CFR100 and General Design Criteria 19). The limit is calculated using the technical specification reactor coolant system (RCS) iodine activity. Application of the circumferential crack repair limit requires the projection of the postulated MSLB leakage based on the projected EOC distribution for the next cycle. The projected EOC distribution is developed using the most recent EOC eddy current results based on crack arc length.

The reduction in the leak rate limit reduces the possibility that a defect in a leaking tube will grow to a size that is not structurally acceptable.

Therefore, this change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

Criterion 2—Does not Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated.

Implementation of the proposed circumferential crack repair limit does not introduce any significant changes to the plant design basis. The only accident possible from implementation of this limit is a tube rupture, which has already been evaluated in the ANO–2 Safety Analysis Report.

The maximum primary-to-secondary leakage rate has been reduced to 150 gallons per day through any one steam generator to help preclude the potential for excessive leakage during all plant conditions. The RG 1.121 criterion for establishing the operational leak rate limit considers: (1) the detection of a crack before potential tube rupture as a result of faulted plant conditions; (2) the maintenance of a margin to tube rupture of not less than three for normal operating conditions; and (3) that any leakage rate increase will be gradual to provide time for corrective action. The 150 gallon per day limit is intended to provide for leakage detection and plant shutdown in the event of an unexpected crack propagation resulting in excessive leakage.

Steam generator tube integrity is maintained through inservice inspection and primary-to-secondary leakage monitoring. Any tubes exceeding the circumferential crack repair limit are removed from service.

Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

Criterion 3—Does not Involve a Significant Reduction in the Margin of Safety.

The use of the circumferential crack repair limit will maintain steam generator tube integrity commensurate with the criteria of RG 1.121. Upon implementation of the limit, even under worst case conditions, the occurrence of circumferential cracking in the expansion transition region is not expected to lead to a steam generator tube rupture event during normal or faulted plant conditions. The distribution of crack indications left in service will result in acceptable primary-to-secondary leakage and conditional tube burst probability during all plant conditions.

The installation of steam generator tube plugs and sleeves reduces RCS flow margin. Implementation of the circumferential crack repair limit will decrease the number of tubes which must be repaired by plugging or sleeving, thereby retaining additional flow margin that would otherwise be reduced.

Therefore, this change does *not* involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Tomlinson Library, Arkansas Tech University, Russellville, AR 72801.

Attorney for licensee: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, N.W., Washington, DC 20005–3502.

NRC Project Director: William D. Beckner.

Florida Power and Light Company, Docket Nos. 50–250 and 50–251, Turkey Point Plant Units 3 and 4, Dade County, Florida

Date of amendment request: May 5, 1995, as supplemented September 28, 1995.

Description of amendment request: The licensee proposes to change Turkey Point Units 3 and 4 Technical Specifications (TS) by revising TS 2.1.1, Safety Limit—Reactor Core; TS 2.2, Limiting Safety System Settings—Reactor Trip System Instrumentation Setpoints; TS 3/4.2.5 Power Distribution Limits— Departure from Nucleate Boiling (DNB) Parameters; TS 3/4.3.2 Engineered Safety Features Actuation System Instrumentation and the associated BASES. The proposed revision to the TS includes (a) the implementation of Westinghouse's NRC approved Revised Thermal Design Procedure (RTDP), and (b) a revision to the Steam Generator Water Level Low-Low trip setpoint.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below. The licensee's analysis was presented separately for the following areas: core thermal limits, overtemperature [delta] T and overpower [delta] T reactor trip setpoint; steam generator process measurement accuracy; and DNB parameter surveillance requirements.

Core Thermal Limits, overtemperature [delta] T and overpower [delta] T Reactor Trip Setpoint

(1) Operation of the facility in accordance with the proposed amendments would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The revised Overtemperature and Overpower [delta] T reactor trip functions do not involve an increase in the probability or consequences of an accident previously evaluated because operation with these revised values will not cause any design or analysis acceptance criteria to be exceeded. The structural and functional integrity of all plant systems is unaffected. The Overtemperature and Overpower [delta] T reactor trip functions are part of the accident mitigation response and are not initiators for any transient. Therefore, the probability of occurrence previously evaluated are not affected.

The changes to the Overtemperature and Overpower [delta] T reactor trip functions do not affect the integrity of the fission product barriers utilized for mitigation of radiological dose consequences as a result of an accident. In addition, the off-site mass releases used as input to the dose calculations are unchanged from those previously assumed. Therefore, the off-site dose predictions remain within the acceptance criteria of 10 CFR Part 100 limits for each of the transients affected. Since it has been concluded that the transient analyses results are unaffected by the parameter modifications, it is concluded that the probability or consequences of an accident previously evaluated are not increased.

(2) The proposed license amendments do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The revised Overtemperature and Overpower [delta] T reactor trip functions do not create the possibility of a new or different kind of accident from any accident previously evaluated because the setpoint adjustments do not affect accident initiation sequences. No new operating configuration is being imposed by the setpoint adjustments that would create a new failure scenario. In addition, no new failure modes or limiting single failures have been identified. Therefore, the types of accidents defined in the UFSAR continue to represent the credible spectrum of events to be analyzed which determine safe plant operation. Therefore, it is concluded that no new or different kind of accidents from those previously evaluated have been created as a result of these revisions.

(3) The proposed license amendments do not involve a significant reduction in a margin of safety.

The changes to the Overtemperature and Overpower [delta] T reactor trip functions do not involve a reduction in the margin of safety because the margin of safety associated with the Overtemperature and Overpower [delta] T reactor trip functions, as verified by the results of the accident analyses, are within acceptable limits. All transients impacted by implementation of the RTDP methodology have been analyzed and have met the applicable accident analyses acceptance criteria. The margin of safety required for each affected safety analysis is maintained. This conclusion is not changed by the Overtemperature and Overpower [delta] T setpoint modifications. The adequacy of the revised Technical Specifications values to maintain the plant in a safe operating condition has been confirmed. Therefore, the changes to the Overtemperature and Overpower [delta] T reactor trip functions do not involve a significant reduction in the margin of safety.

Steam Generator Process Measurement Accuracy

(1) Operation of the facility in accordance with the proposed amendments would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The revised reactor trip setpoints on Steam Generator water level do not involve a significant increase in the probability or consequences of an accident previously evaluated. Operation with these revised values will not cause any design or analysis acceptance criteria to be exceeded. The structural and functional integrity of any plant system is unaffected. The Steam Generator Water Level trip functions are part of the accident mitigation response and are not themselves initiators for any transient. Therefore, the probability of occurrence previously evaluated is not affected.

The changes to the reactor trip setpoints do not affect the integrity of the fission product barriers utilized for mitigation of radiological dose consequences as a result of an accident. The Steam Generator Water Level Low-Low trip setpoint assumed in the safety analyses has been revised and acceptable results were obtained. The Steam Generator Water Level-Low setpoint is not credited in the safety analysis. Consequently, the required margin of safety for each affected safety analysis has been maintained. In addition, the offsite mass releases used as input to the dose calculations are unchanged from those previously assumed. Therefore, the offsite dose predictions remain within the acceptance criteria of 10 CFR Part 100 limits for each of the transient analyses affected. Since it has been determined that the transient analysis results are unaffected by these parameter modifications, FPL concludes that the consequences of an accident previously evaluated are not increased.

(2) The proposed license amendments do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The setpoint values do not affect the assumed accident initiation sequences. In addition, no new failure modes or limiting single failures have been identified for any plant equipment. Therefore, the types of accidents defined in the UFSAR continue to represent the credible spectrum of events to be analyzed which determine safe plant operation. Therefore, the possibility of a new or different kind of accident from any accident evaluated is not increased.

(3) The proposed license amendments do not involve a significant reduction in the margin to safety.

The current Technical Specification trip setpoints and allowable values were changed to maintain the current safety analysis limits. The Steam Generator Water Level Low-Low trip setpoint assumed in the safety analyses has been revised and acceptable results were obtained. The Steam Generator Water Level-Low setpoint is not credited in the safety analysis. Consequently, the required margin of safety for each affected safety analysis has been maintained. Thereby, the adequacy of the revised Technical Specification values to maintain the plant in a safe operating condition is also confirmed.

DNB Parameter Surveillance Requirements

(1) Operation of the facility in accordance with the proposed amendments would not involve a significant increase in the probability or consequences of an accident previously evaluated.

With the retention of the previous Safety Analyses Limits for Departure from Nucleate Boiling (DNB) (T.S. 3/4.2.5) and the existing Reactor Coolant System (RCS) low flow trip Nominal Trip Setpoint (NTS), there is no increase in the probability or consequences of an accident previously evaluated because there is no change to any design or analysis acceptance criteria. The structural and functional integrity of any plant system is unaffected. The proposed license amendments revise the surveillance requirements for DNB parameters and incorporate the RTDP uncertainty analysis into the Westinghouse methodology for the RCS Loss of Flow determination of the Allowable Value.

The changes to the reactor trip functions do not affect the integrity of the fission product barriers utilized for mitigation of radiological dose consequences as a result of an accident. The margin to safety for the RCS Loss of Flow trip remains protected as the trip setpoints assumed in the safety analyses are not revised. In addition, the offsite mass releases used as input to the dose calculations are unchanged from those previously assumed. Therefore, the offsite dose predictions remain within the acceptance criteria of 10 CFR Part 100 limits for each of the transients affected. Since it has been determined that the transient results are unaffected by these parameter modifications, it is concluded that the consequences of an accident previously evaluated are not increased.

(2) Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The revised Allowable Value does not create the possibility of a new or different kind of accident from any accident previously evaluated. Revision of the surveillance requirements merely provides clarification to more accurately reflect the surveillance activity.

The Allowable Value does not affect the assumed accident initiation sequences. In addition, no new failure modes or single failures have been identified for any plant equipment. Therefore, the types of accidents defined in the UFSAR continue to represent the credible spectrum of events to be analyzed which determine safe plant operation. Therefore, it is concluded that no new or different kind of accidents from those previously evaluated have been created as a result of these revisions.

(3) The proposed license amendments do not involve a significant reduction in the margin to safety.

The RCS Loss of Flow setpoint assumed in the safety analysis remains unchanged. Since the safety analysis limit setpoint value is unchanged and no safety analysis is affected, the required margin of safety for each affected safety analysis is maintained. Thereby, the adequacy of the revised Technical Specification values to maintain the plant in a safe operating condition is also confirmed. Therefore, the change to the RCS Loss of Flow Allowable Value does not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Florida International University, University Park, Miami, Florida 33199.

Attorney for licensee: J. R. Newman, Esquire, Morgan, Lewis & Bockius, 1800 M Street, NW., Washington, DC 20036.

NRC Project Director: David B. Matthews.

Florida Power and Light Company, Docket Nos. 50–250 and 50–251, Turkey Point Plant Units 3 and 4, Dade County, Florida

Date of amendment request: July 26, 1995, as supplemented by letter dated October 4, 1995.

Description of amendment request: The licensee proposes to revise the technical specifications surveillance intervals and allowed outage times for the channel operational tests performed on the analog ''bistable'' comparator modules for the reactor trip, reactor trip permissive functions, engineered safety features actuation and permissive functions identified below.

TS Table 3.3–1—Revise ACTION Statements 2a, 6, 12 and 13; increase the time allowed for a channel to be inoperable or out of service in an untripped condition from 1 hour to 6 hours. Revise ACTION Statement 2b; increase the time a Nuclear Instrumentation System (NIS) channel in a functional group may be bypassed to perform testing from 2 to 4 hours.

TS Table 3.3–2—Revise ACTION Statement 14; increase the time to be in HOT STANDBY with the number of OPERABLE channels one less that the Minimum Channels OPERABLE requirement from 6 to 12 hours. Revise ACTION Statements 14, 20 and 22; increase the allowed outage time for test of the logic trains from 2 hours to 8 hours. Revise ACTION Statements 15, 18 and 25; increase the time allowed for a channel to be inoperable and out of service in an untripped condition from 1 hour to 6 hours.

TS Table 4.3–1—Revise the surveillance interval for Items 2.a, 4, 7, 8, 10, 11, 12 and Note (9) from monthly to quarterly. Revise the surveillance interval for Item 2.b from monthly to startup, and Item 3 from monthly/startup to startup only. Revise the surveillance interval for Items 17.a, 17.b, 17.c and 17.d from monthly to refueling. Revise Note (1) from ''7 days'' to ''31 days'' and delete Note (8).

TS Table 4.3–2—Revise the surveillance interval for Items 1.d, 1.e, 1.f, 4.d, 5.c, 6.b, and 8.a from monthly to quarterly.

TS BASES 3/4.3.1 and 3/4.3.2—Revise the BASES section for Technical Specification Sections 3/4.3.1 and 3/4.3.2 to reference the Westinghouse WCAPs 10271 and 10271, Supplement 2, and associated Nuclear Regulatory Commission (NRC) safety evaluation reports (SERs).

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

(1) Operation of the facility in accordance with the proposed amendments would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes in Technical Specification surveillance intervals and allowed outage times for the subject Reactor Protection System (RPS)/Nuclear Instrumentation System (NIS)/Engineered Safety Features Actuation System (ESFAS) analog instrumentation have been revised in accordance with the recommendations and criteria of Westinghouse WCAP–10271, WCAP 10271, Supplement 2, and the NRC's SERs on the same subject dated February 21, 1985 and dated February 22, 1989.

The proposed changes do not involve any hardware or setpoint changes. Similarly, the proposed changes do not alter the manner in which safety limits, limiting safety system setpoints or limiting conditions for operation are determined. Implementation of the proposed changes does affect the probability of failure of the RPS, including NIS, and ESFAS, but does not alter the manner in which protection is afforded nor the manner in which limiting setpoint criteria are established for the RPS/ESFAS instrumentation systems. Consequently, the proposed changes do not result in an increase in the severity or consequences of any accident previously evaluated.

Implementation of the proposed changes is expected to result in an acceptably small increase in total RPS unavailability. This increase is primarily due to less frequent surveillances and was generically quantified to be less than 3% within WCAP–10271. WCAP–10271 also documents that the implementation of the proposed changes is also expected to result in a significant reduction in the probability of core melt from inadvertent reactor trips (WCAP–10271). This is the result of a reduction in the number of inadvertent reactor trips (0.5 fewer inadvertent reactor trips per unit per year) occurring during testing of the RPS instrumentation. This reduction is primarily attributable to testing in bypass for applicable channels and to less frequent surveillances. WCAP–10271 documents that the reduction in inadvertent core melt probability is sufficiently large to counter the increased core melt probability, resulting in an overall reduction in total core melt probability of approximately 1%.

A corresponding probabilistic risk assessment (WCAP–10271, Supplement 2) was documented by Westinghouse for the generic implementation of the proposed changes for ESFAS instrumentation. This Westinghouse evaluation along with the independent assessments performed by an NRC contractor demonstrated that a 6% core damage frequency increase represented an upper bound for Westinghouse plants. For more realistic testing strategies, the core damage frequency increase would be substantially less than this.

Consequently, the changes in Technical Specifications associated with an extension of the surveillance intervals and out of service times for the RPS/ESFAS instrumentation systems will have only a small impact on plant risk. On this basis, FPL concludes that the proposed changes will not have a significant effect on the probability or consequences of licensing basis events; and the probability or consequences of an accident previously evaluated for Turkey Point does not significantly increase.

(2) Operation of the facility in accordance with the proposed amendments would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes in Technical Specification surveillance intervals and allowed outage times for the subject RPS/ESFAS analog instrumentation have been revised in accordance with the recommendations and criteria of Westinghouse WCAP–10271, WCAP 10271, Supplement 2, and the NRC's SERs on the same subject dated February 21, 1985 and dated February 22, 1989.

The proposed changes do not involve any hardware or setpoint changes. Some existing instrumentation is designed to be tested in bypass and current Technical Specifications allow testing in bypass. Testing in bypass is also recognized by IEEE Standards.

Therefore, testing in bypass has been previously approved and implementation of the proposed changes for testing in bypass does not create the possibility of a new or different kind of accident from any previously evaluated. Furthermore, since the proposed changes do not alter the manner in which protection is afforded nor the manner in which limiting criteria are established for the RPS and ESFAS instrumentation systems, the possibility of a new or different kind of accident from any previously evaluated has not been created.

The proposed changes do not result in a change in the manner in which the RPS or ESFAS provides plant protection. No change is being made which alters the function of the RPS or ESFAS (other than in a test mode). Rather, the likelihood or probability of the RPS and ESFAS functioning properly is the only effect.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident nor involve a reduction in a margin of safety as defined in the Safety Analysis Report.

Consequently, the changes in Technical Specifications associated with an extension of the surveillance intervals and out of service times for the RPS/ESFAS instrumentation systems will not create the possibility of a new or different kind of accident from any previously evaluated by the NRC, and does not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Operation of the facility in accordance with the proposed amendments would not involve a significant reduction in a margin of safety.

The proposed changes in Technical Specification surveillance intervals and allowed outage times for the subject RPS/ESFAS analog instrumentation have been revised in accordance with the recommendations and criteria of Westinghouse WCAP–10271, WCAP 10271, Supplement 2, and the NRC's SERs on the same subject dated February 21, 1985 and dated February 22, 1989.

These changes in Technical Specifications only affect the frequency of the channel operational tests and the allowed outage times; they do not alter the manner in which protection is afforded nor the manner in which limiting setpoint criteria are established. In addition, the fundamental process to implement these channel operational tests remains the same.

The proposed changes do not alter the manner in which safety limits, limiting safety system setpoints or limiting conditions for operation are determined. The impact of reduced testing is to allow a longer time interval over which instrument uncertainties (e.g., drift) may act. The site specific review of historical drift data and the conservative application of drift in the Westinghouse methodology are sufficient to demonstrate that the basis of the Technical Specification setpoint determinations are not adversely affected by extending the surveillance interval from monthly to quarterly, that is, quarterly surveillance test intervals would not exceed the allowable instrument drift of these analog devices.

Implementation of the proposed changes is expected to result in an overall improvement in safety by:

(a) Fewer inadvertent reactor trips per unit per year. This is due to less frequent testing which minimizes the time spent in a partial trip condition.

(b) Higher quality repairs leading to improved equipment reliability due to longer allowed repair times.

(c) Improvements in the effectiveness of the operating staff in monitoring and controlling plant operation. This is due to less frequent distractions of the operator and shift supervisor from attending to instrumentation testing.

The Westinghouse analysis demonstrates that any expected increases in probability of core melt or core damage frequency are small and are therefore acceptable. Consequently, the changes in Technical Specifications associated with an extension of the surveillance intervals and out of service times for the RPS/ESFAS instrumentation systems will not significantly reduce the margin of plant safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Florida International University, University Park, Miami, Florida 33199.

Attorney for licensee: J.R. Newman, Esquire, Morgan, Lewis & Bockius, 1800 M Street NW., Washington, DC 20036.

NRC Project Director: David B. Matthews.

Maine Yankee Atomic Power Company, Docket No. 50–309, Maine Yankee Atomic Power Station, Lincoln County, Maine

Date of amendment request: July 24, 1995

Description of amendment request: The proposed amendment would modify Technical Specification (TS) 3.12.B by adding an Exception to permit a once-per-operating cycle 10 day restoration time for Remedial Action statement 3.12.B.2. The extended restoration time would allow maintenance to be completed on the emergency diesel generators. In addition, the Basis of TS 3.12 is supplemented in support of the proposed amendment.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration. The NRC staff has reviewed the licensee's analysis against the standards of 10 CFR 50.92(c). The NRC staff's review is presented below:

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The emergency diesel generators (EDG) are not accident initiators for any accident previously evaluated, nor does the proposed change affect any of the assumptions used in the deterministic safety analyses. To evaluate the effect of the proposed extended restoration time of the EDGs fully, probabilistic safety analysis (PSA) methods were used. The results of these analyses show no significant increase in core damage frequency. Thus, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change does not alter the design, configuration, or method of operation of the plant. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

The proposed change does not affect system or component limiting conditions for operation, or the bases used in the deterministic analyses to establish the margin of safety. The PSA evaluations used to evaluate the proposed change demonstrated that the changes are either risk neutral or risk beneficial. Thus the proposed change does not involve a significant reduction in a margin of safety.

Based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Wiscasset Public Library, High Street, P.O. Box 367, Wiscasset, ME 04578.

Attorney for licensee: Mary Ann Lynch, Esquire, Maine Yankee Atomic Power Company, 329 Bath Road, Brunswick, ME 04011.

NRC Project Director: Phillip F. McKee.

Maine Yankee Atomic Power Company, Docket No. 50–309, Maine Yankee Atomic Power Station, Lincoln County, Maine

Date of amendment request: August 15, 1995.

Description of amendment request: The proposed amendment would allow reduced power operation as a function of total reactor coolant flow, for flow reductions as much as 5 percent below the currently specified minimum flow. Specifically, operation would be allowed with total flow rates below 360,000 gpm, if rated thermal power is reduced by 1.5 percent for each 1.0 percent that total reactor coolant flow is reduced.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration. The NRC staff has reviewed the licensee's analysis against the standards of 10 CFR 50.92(c). The staff's review is presented below:

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed amendment does not involve any changes in the configuration of the reactor coolant system. Thus, precursors to accidents previously evaluated are unchanged. The 5.0 percent reduction in reactor coolant flow introduces a relatively minor change to the overall plant heat balance, which is conservatively offset by the proposed requirement to reduce rated thermal power by 1.5 percent for each 1.0 percent reduction in reactor coolant system flow. Analysis by the licensee shows that a 1.0 percent reduction in rated thermal power for every 1.0 percent reduction in reactor coolant system flow is sufficient to ensure that the current departure from nuclear boiling ratio is maintained. The licensee asserts that achieving the reduced power and other, related limits, within 24-hours of a subject flow reduction will not significantly increase the probability or consequences of an accident previously evaluated. Thus, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed amendment does not involve any modifications or additions to plant equipment, and the design and operation of the plant are not affected. The reduction in rated thermal power, reactor protection system trip points, and operating limits conservatively offset the reduction in reactor coolant system flow. Plant operating conditions remain bounded by Final Safety Analysis Report (FSAR) Chapter 14, Safety Analysis. Thus, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

Plant rated power is conservatively reduced, consistent with the reactor coolant flow reduction. The power reduction is specifically designed to maintain the margin to the specified acceptable fuel design limit on the departure from nuclear boiling ratio (DNBR), as defined in MY TS 2.2. The licensee has evaluated this margin using the methodologies identified in Maine Yankee Technical Specification 5.14. The reduction in power level, operating limits, and reactor protection system setpoints ensures that the DNBR margin is maintained for those FSAR Chapter 14 events that rely on automatic reactor trip protection. Power level reductions ensure that the total sensible heat in the reactor coolant system is conservative for those events dependent on initial system energy. Thus, the proposed change does not involve a significant reduction in a margin of safety.

Based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that this amendment request involves no significant hazards consideration.

Local Public Document Room location: Wiscasset Public Library, High Street, P.O. Box 367, Wiscasset, ME 04578.

Attorney for licensee: Mary Ann Lynch, Esquire, Maine Yankee Atomic Power Company, 329 Bath Road, Brunswick, ME 04011.

NRC Project Director: Phillip F. McKee.

Northeast Nuclear Energy Company, et al., Docket No. 50–336, Millstone Nuclear Power Station, Unit Nos. 2, New London, Connecticut

Date of amendment request: September 19, 1995.

Description of amendment request: The proposed amendment would reduce the frequency of the surveillance interval of the Safety Injection Tanks (SITs) boron concentration from once per 31 days to once per 6 months.

Basis for proposed no significant hazards consideration determination:

As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration (SHC), which is presented below:

Pursuant to 10CFR50.92, Northeast Nuclear Energy Company (NNECO) has reviewed the proposed change. NNECO concludes that the change does not involve a significant hazards consideration since the proposed change satisfies the criteria in 10CFR50.92(c). That is, the proposed change does not:

1. Involve a significant increase in the probability or consequences of an accident previously analyzed.

The revised Safety Injection Tank (SIT) surveillance requirements meet all design and performance criteria. The change has no [e]ffect on the ability of the SIT to perform its designed function of providing borated water to the core following a depressurization as a result of a Loss of Coolant Accident (LOCA). Therefore, the changes to SIT surveillance requirements will not increase the probability or consequences of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any previously analyzed.

The revised SIT surveillance requirements meet all design and performance criteria. The change has no [e]ffect on the ability of the SIT to perform its design function of providing borated water to the core following a depressurization as a result of a LOCA. The change to the SIT surveillance requirement will not create the possibility of a new or different kind of accident from any previously analyzed.

3. Involve a significant reduction in the margin of safety.

The boron concentration of the SIT will not be affected by the change to the surveillance requirement. The boron concentration within the SIT will continue to be monitored on a basis consistent with the historical performance. These changes will have no impact on the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, CT 06360.

Attorney for licensee: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, CT 06141–0270.

NRC Project Director: Phillip F. McKee.

Northeast Nuclear Energy Company, et al., Docket No. 50–336, Millstone Nuclear Power Station, Unit Nos. 2, New London, Connecticut

Date of amendment request: September 29, 1995.

Description of amendment request: The proposed amendment would modify the Technical Specifications 3.4.2.1, 3.4.2.2, 3.7.1.1, and Table 4.7–1.

The proposed license amendment combines three separate changes to the Millstone Unit No. 2 Technical Specifications which pertain to safety valves. The first proposed modification would expand the as-found tolerance of the lift setting pressure for the pressurizer and the main steam safety valves from the current value of plus or minus 1 percent to plus or minus 3 percent. Clarifications have also been proposed by specifying that the lift setting pressure shall be determined at normal operating conditions and shall be set within plus or minus 1 percent of the required lift setting. The second portion of the modification would eliminate the need to verify the main steam safety valve orifice size. The third modification would modify the main steam safety valve action statement to reflect that if a main steam safety valve is inoperable and compensating action cannot be taken that the plant must be brought to hot shutdown (Mode 4) in 12 hours instead of cold shutdown (Mode 5) in 30 hours.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration (SHC), which is presented below:

* * * The proposed changes do not involve an SHC because the changes would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

The change in the as-found pressurizer safety valve tolerance will not increase the probability of occurrence of any of the design basis accidents. Even with the larger tolerance, the setpoint will provide margin to normal operation, the reactor setpoint, and PORV [power-operated relief valve setpoint]. This minimizes the challenges to safety valves and assures that there is no increase in the probability of an inadvertent opening of a pressurizer safety valve. Similarly, even with the increase in allowed as-found tolerance for the main steam safety valves, the setpoints will still provide margin to normal operation. Thus, there is no impact on the probability of an inadvertent opening of a steam generator safety valve.

The loss of load event and the inadvertent closure of one main steam isolation valve have been reanalyzed to show that even with a [plus or minus] 3 percent tolerance for the pressurizer safety valves and the main steam safety valves, that both the peak RCS [reactor coolant system] pressure and the peak steam generator pressure remain below 110 percent of design. Thus, even with the larger as-found tolerances, the margin of safety for RCS and steam generator overpressurization is maintained.

The steam generator tube rupture has been reanalyzed to take into account the [plus or minus] 3 percent as-found tolerance and to extend the margin for operator action to one hour. A comparison of the calculated doses shows that with the new assumptions, there would be a very small increase in calculated doses. The increased calculated doses, however, remain well below the Standard Review Plan acceptance criteria.

The proposed change in the shutdown mode does not impact the probability or consequences of an accident previously evaluated. The proposed change makes the action required for inoperable main steam safety valves consistent with the modes that the technical specification is applicable and would not modify the assumptions made in any accident previously analyzed.

The change to delete the main steam safety valve orifice size from technical specifications has no impact on any design basis accident analysis.

Based upon these evaluations, it is concluded that the proposed changes do not significantly increase the probability or consequences of any design basis accident.

2. Create the possibility of a new or different kind of accident from any previously analyzed.

The proposed changes do not create the possibility of a new or different kind of accident from any previously analyzed.

The proposed changes do not change the as-left setpoints. The change in as-found tolerances for the safety valves is being made to reflect the results of past surveillances that indicate that the setpoints can drift more than the current criteria. However, there is no change in the plant configuration or in as-left setpoints.

The proposed change which requires the plant to go to Mode 4 in 12 hours instead of Mode 5 in 30 hours if the action statement is not met, is consistent with the applicable modes of the technical specification (i.e., the technical specification is not applicable in Mode 4). No new or different kind of accident from those previously analyzed can be postulated as a result of this proposed change.

Thus, the changes do not create the possibility of a new or different kind of accident from any previously analyzed.

3. Involve a significant reduction in the margin of safety.

As discussed above, the loss of load event and the inadvertent closure of one main steam isolation valve have been reanalyzed to show that even with a [plus or minus] 3 percent tolerance for the pressurizer safety valves and the main steam safety valves, that both the peak RCS pressure and the peak steam generator pressure remain below 110 percent of design. Thus, even with the larger as-found tolerances, the margin of safety for RCS and steam generator overpressurization is maintained. In addition, the steam generator tube rupture has been reanalyzed with a [plus or minus] 3 percent tolerance on the steam generator safety valves and the results show an insignificant increase in the calculated doses.

The proposed change also directs the operator to bring the plant to hot shutdown instead of cold shutdown to be consistent with the applicable modes of the technical specification. There is no impact on the assumptions made or the results of any accident previously analyzed.

Therefore, it is concluded that the changes do not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, CT 06360.

Attorney for licensee: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, CT 06141–0270.

NRC Project Director: Phillip F. McKee

Pennsylvania Power and Light Company, Docket Nos. 50–387 and 50–388 Susquehanna Steam Electric Station, Units 1 and 2, Luzerne County, Pennsylvania

Date of amendment request: September 18, 1995

Description of amendment request: The proposed amendment would relocate Fire Protection requirements from the Technical Specifications to the Technical Requirements Manual. In addition, the proposed amendment would revise Technical Specifications to include the requirement for a program and procedure to implement the Technical Requirements Program, and also revises Technical Specifications to add the requirement for the Plant Operations Review Committee to review all proposed changes to the Technical Requirements Program and to forward copies of reviewed changes to the Susquehanna Review Committee.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

The proposed changes do not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change relocates the provisions of the Fire Protection Program that are contained in the Technical Specifications and places them in the Technical Requirements Manual. No requirements are being added or deleted. Review and approval of those portions of the Fire Protection Program contained in the Technical Requirements Manual and revisions thereto will be the responsibility of the Plant Operations Review Committee just as it was their responsibility to review changes to the fire protection Limiting Condition for Operation and Surveillance Requirements when they were part of the Technical Specifications. Requiring review by the Plant Operations Review Committee reinforces the importance of the Technical Requirements Manual and the requirements controlled by it and assures a multidisciplined review. Approved Technical Requirements or changes thereto are provided to the Susquehanna Review Committee for information. No design basis accidents are affected by the change, nor are safety systems adversely affected by the change. Therefore, there is no impact on the probability of concurrence [occurrence] or the consequences of any design basis accidents.

2. Create the possibility of a new or different kind of accident from any previously evaluated.

The proposed changes relocate the provisions of the Fire Protection Program that are contained in the Technical Specifications and places them in the Technical Requirements Manual. No requirements are being added or deleted by the Technical Requirements Manual. There are no new failure modes associated with the proposed changes. Therefore, since the plant will continue to operate as designed, the proposed changes will not modify the plant response to an accident.

3. Involve a significant reduction in a margin of safety.

No change is being proposed for the Fire Protection Program requirements themselves. The relevant Technical Specifications are being relocated, and the requirements contained therein are being incorporated into the Technical Requirements Manual. Plant procedures will continue to provide the specific instructions necessary for the implementation of the requirements, just as when the requirements resided in the Technical Specifications. Fire Protection Program changes will be subject to the provisions of 10 CFR 50.59 and the current fire protection license condition. As such, the changes do not directly affect any protective boundaries nor does it [do they] impact the safety limits for the boundary. Review and approval of those portions of the Fire Protection Program contained in the Technical Requirements Manual and the revisions thereto will be the responsibility of the Plant Operations Review Committee just as it was their responsibility to review changes to the fire protection Limiting Condition for Operation and Surveillance Requirements when they were part of the Technical Specification. Approved Technical Requirements or changes thereto are provided to the Susquehanna Review Committee for information. Thus, there are no adverse impacts on the protective boundaries, safety limits, or margin of safety.

Since operability and surveillance requirements will remain in a controlled document, the changes do not reduce the effectiveness of Technical Specification requirements. Any changes to the Fire Protection Program requirements will be made in accordance with the provisions of 10 CFR 50.59 and the fire protection license condition.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Osterhout Free Library, Reference Department, 71 South Franklin Street, Wilkes-Barre, Pennsylvania 18701.

Attorney for licensee: Jay Silberg, Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street NW., Washington, DC 20037.

NRC Project Director: John F. Stolz.

Virginia Electric and Power Company, Docket No. 50–338, North Anna Power Station, Unit No. 1, Louisa County, Virginia

Date of amendment request: September 19, 1995.

Description of amendment request: The proposed change would revise the Technical Specifications (TS) for the North Anna Power Station, Units 1 & 2 (NA–1 & 2). Specifically, the proposed changes would revise TS Limiting Condition for Operation (LCO) 3.7.1.1 Action Statements, TS Table 3.7–1, dually entitled ''Maximum Allowable Power Range Neutron Flux High Setpoint With Inoperable Steam Line Safety Valves During 3 Loop Operation'' and ''Maximum Allowable Power Range Neutron Flux High Setpoint With Inoperable Steam Line Safety Valves During 2 Loop Operation,'' and the TS Bases 3/4.7.1.1, ''Safety Valves'' for NA–1 & 2. Table 3.7–1 provides the maximum allowable power range neutron flux high setpoints with one or more main steam safety valves (MSSVs) inoperable during two loop and three loop operation. The proposed changes provide more conservative power range neutron flux high setpoints calculated utilizing the Westinghouse Electric Corporation (Westinghouse) recommended methodology and delete the information for setpoints for two loop operation. The proposed changes also revise the TS Bases to reflect the methodology used to establish the new setpoints, and delete the LCO Action Statement and the TS Bases for two loop operation.

Additionally, the information in Table 3.7–1 and the LCO Action Statement associated with two loop operation have been deleted since Virginia Electric and Power Company is prohibited by the license from operating in this configuration.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

Specifically, operation of the North Anna Power Station in accordance with the proposed Technical Specifications changes will not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

This change reduces the power level at which the reactor may be operated with one or more main steam safety valves (MSSVs) inoperable to ensure that the secondary system is not overpressurized during the most severe pressurization transient of the secondary side. There is no change to the function of the MSSVs by the proposed change and will not alter any accident analysis assumptions or results. The proposed changes will provide conservative power range neutron flux high trip setpoints such that the maximum power level allowed for operation with inoperable MSSVs is below the heat removing capability of the operable MSSVs. Therefore, this change will not increase the probability of an accident.

This change is consistent with the current accident analysis assumptions for the MSSVs and does not change the containment response for any design basis event. Therefore, no change in the mitigation of an accident will result from this proposed change and no change will occur in the consequences of any accident currently analyzed.

2. Create the possibility of a new or different kind of accident from any accident previous[ly] evaluated.

Since the implementation of the proposed changes to the setpoints will not require hardware modifications (i.e., alterations to plant configuration), operation of the facilities with these proposed Technical Specifications does not create the possibility for any new or different kind of accident which has not already been evaluated.

The proposed revision to the Technical Specifications will not result in any physical alteration to any plant system, nor would there be a change in the method by which any safety-related system performs its function. The design and operation of the main steam system is not being changed.

These changes do not change the design, operation, or failure modes of the main steam system. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Involve a significant reduction in a margin of safety.

The proposed change reduces the total energy of the reactor coolant system that will ensure the ability of the MSSVs to perform their intended function as assumed in the current accident analyses. Correcting this non conservatism restores the margin of safety to what was originally envisioned. In addition, the results of the accident analyses which are documented in the UFSAR bound operation under the proposed changes, so that there is no safety margin reduction. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: The Alderman Library, Special Collections Department, University of Virginia, Charlottesville, Virginia 22903–2498.

Attorney for licensee: Michael W. Maupin, Esq., Riverfront Plaza, East Tower, 951 E. Byrd Street, Richmond, Virginia 23219.

NRC Project Director: David B. Matthews.

Virginia Electric and Power Company, Docket Nos. 50–338 and 50–339, North Anna Power Station, Units No. 1 and No. 2, Louisa County, Virginia

Date of amendment request: September 19, 1995.

Description of amendment request: The proposed change would revise the Technical Specifications (TS) for the North Anna Power Station, Units No. 1 and No. 2 (NA–1&2). Specifically, the proposed change would increase the surveillance test interval for the turbine reheat stop and intercept valves to once per 18 months and extend the visual and surface inspection interval to 60 months. The proposed change would also remove the requirement to perform additional visual and surface inspections on the remaining turbine overspeed protection system control valves of that type when unacceptable flaws or excessive corrosion are identified which can be directly attributed to a service condition specific to the inspected valve.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

Specifically, operation of the North Anna Power Station in accordance with the proposed Technical Specifications changes will not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

No new or unique accident precursors are introduced by these changes in surveillance requirements. The probability of turbine missile ejection with an extended 18-month test interval for the reheat stop and intercept valves has been determined to be within the applicable acceptance criteria.

The heavy hub design of the turbine rotors provides further assurance that the probability of the ejection of destructive missiles remains minimal.

Based upon the results of the probabilistic evaluation, the probability of a turbine generated missile is less than 10^{-5} per year which the Commission has endorsed as the acceptable level for turbine operation.

The reheat stop and intercept valve inspection interval extension and the elimination of the additional visual/surface inspections do not change the design, operation, or failure modes of the valves and other components in the turbine overspeed protection system.

Therefore, these changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

The demonstrated high reliability of the turbine reheat stop and intercept valves and the verification of the operability of the other turbine control valves provide adequate assurance that the turbine overspeed protection system will operate as designed, if needed. Turbine reheat stop and intercept valve testing performed to date has demonstrated the reliability of these valves. In addition, the operability of the other turbine valves (i.e., turbine throttle valves and governor valves) will continue to be verified every 31 days or as required by the Technical Specifications.

2. Create the possibility of a new or different kind of accident from any accident previous[ly] evaluated.

Since the implementation of the proposed change to the surveillance requirements will not require hardware modifications (i.e., alterations to plant configuration), operation of the facilities with these proposed Technical Specifications does not create the possibility for any new or different kind of accident which has not already been evaluated in the Updated Final Safety Analysis Report (UFSAR). In addition, the results of the probabilistic evaluation indicate that no additional transients have been introduced.

The proposed revision to the Technical Specifications will not result in any physical alteration to any plant system, nor would there be a change in the method by which any safety-related system performs its function. The design and operation of the turbine overspeed protection and turbine control systems are not being changed.

The proposed Technical Specifications changes do not affect the design, operation, or failure modes of the valves and other components of the turbine overspeed protection system.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Involve a significant reduction in a margin of safety.

The proposed changes would not reduce the margin of safety as defined in the basis for any Technical Specifications. The design and operation of the turbine overspeed protection and turbine control systems are not being changed and the operability of the turbine reheat stop and intercept valves will be demonstrated on a refueling outage basis. In addition, the results of the accident analyses which are documented in the UFSAR continue to bound operation under the proposed changes, so that there is no safety margin reduction. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: The Alderman Library, Special Collections Department, University of Virginia, Charlottesville, Virginia 22903–2498.

Attorney for licensee: Michael W. Maupin, Esq., Hunton and Williams, Riverfront Plaza, East Tower, 951 E. Byrd Street, Richmond, Virginia 23219.

NRC Project Director: David B. Matthews.

Previously Published Notices of Consideration of Issuance of Amendments to Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing

The following notices were previously published as separate individual notices. The notice content was the same as above. They were published as individual notices either because time did not allow the Commission to wait for this biweekly notice or because the action involved exigent circumstances. They are repeated here because the biweekly notice lists all amendments issued or proposed to be issued involving no significant hazards consideration.

For details, see the individual notice in the Federal Register on the day and page cited. This notice does not extend the notice period of the original notice.

Commonwealth Edison Company, Docket Nos. STN 50–456 and STN 50–457, Braidwood Station, Unit Nos. 1 and 2, Will County, Illinois

Date of amendment request: August 15, 1995.

Description of amendment request: This application to revise the Braidwood, Unit 1, Technical Specifications (TSs) proposes to continue to use the voltage-based repair criteria which were added to the Braidwood, Unit 1, TSs by a license amendment issued on August 18, 1994. This August 15, 1995, request will be considered by the staff only in the event that the staff can not reach a timely decision on your pending request for license amendments dated September 1, 1995, to raise the present lower voltage repair limit from 1.0 volt to 3.0 volts.

Date of publication of individual notice in Federal Register: October 5, 1995 (60 FR 52222).

Expiration date of individual notice: November 6, 1995.

Local Public Document Room location: Wilmington Public Library, 201 S. Kankakee Street, Wilmington, Illinois 60481.

Commonwealth Edison Company, Docket Nos. 50–237 and 50–249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, Illinois

Docket Nos. 50–254 and 50–265, Quad Cities Nuclear Power Station, Units 1 and 2, Rock Island County, Illinois

Date of amendment request: September 15, 1995.

Description of amendment request: To close out open items identified by the NRC staff's review of the upgrade of sections 1.0, 3/4.4, 3/4.10, and 5.0 of the Dresden and Quad Cities Technical Specifications to the BWR Standard Technical Specifications.

Date of publication of individual notice in Federal Register: October 5, 1995 (60 FR 52220).

Expiration date of individual notice: November 6, 1995.

Local Public Document Room location: for Dresden, Morris Area Public Library District, 604 Liberty Street, Morris, Illinois 60450; for Quad Cities, Dixon Public Library, 221 Hennepin Avenue, Dixon, Illinois 61021

Commonwealth Edison Company, Docket Nos. 50–254 and 50–265, Quad Cities Nuclear Power Station, Units 1 and 2, Rock Island County, Illinois

Date of amendment request: September 20, 1995.

Description of amendment request: The proposed amendment would upgrade the Quad Cities TS to the Standard Technical Specifications (STS) contained in NUREG–0123. The Technical Specification Upgrade Program (TSUP) is not a complete adaption of the STS. The TS upgrade focuses on (1) integrating additional information such as equipment operability requirements during shutdown conditions, (2) clarifying requirements such as limiting conditions for operation and action statements utilizing STS terminology, (3) deleting superseded requirements and modifications to the TS based on the licensee's responses to Generic Letters (GL), and (4) relocating specific items to more appropriate TS locations. The September 20, 1995, application proposed to upgrade only Section 6.0 (Administrative Controls) of the Quad Cities TS.

Date of publication of individual notice in Federal Register: October 5, 1995 (60 FR 52226).

Expiration date of individual notice: November 6, 1995.

Local Public Document Room location: Dixon Public Library, 221 Hennepin Avenue, Dixon, Illinois 61021.

North Atlantic Energy Service Corporation, Docket No. 50–443, Seabrook Station, Unit No. 1, Rockingham County, New Hampshire

Date of amendment request: September 5, 1995.

Description of amendment request: The proposed amendment would modify the Appendix A Technical Specifications (TSs) for the Turbine Cycle Safety Valves. Specifically, the proposed amendment would change Seabrook Station Appendix A Technical Specification Table 3.7–1 to reduce the maximum allowable Power Range Neutron Flux—High setpoints with inoperable Main Steam Safety Valves (MSSVs) and Table 3.7–2 to reduce the opening setpoints of the MSSVs.

Date of publication of individual notice in Federal Register: October 2, 1995 (60 FR 51505).

Expiration date of individual notice: November 1, 1995.

Local Public Document Room location: Exeter Public Library, Founders Park, Exeter, NH 03833.

Notice of Issuance of Amendments to Facility Operating Licenses

During the period since publication of the last biweekly notice, the Commission has issued the following amendments. The Commission has determined for each of these amendments that the application complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in

10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment to Facility Operating License, Proposed No Significant Hazards Consideration Determination, and Opportunity for A Hearing in connection with these actions was published in the Federal Register as indicated.

Unless otherwise indicated, the Commission has determined that these amendments satisfy the criteria for categorical exclusion in accordance with 10 CFR 51.22. Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared for these amendments. If the Commission has prepared an environmental assessment under the special circumstances provision in 10 CFR 51.12(b) and has made a determination based on that assessment, it is so indicated.

For further details with respect to the action see (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document rooms for the particular facilities involved.

Arizona Public Service Company, et al., Docket Nos. STN 50–528, STN 50–529, and STN 50–530, Palo Verde Nuclear Generating Station, Units 1, 2, and 3, Maricopa County, Arizona

Date of application for amendments: August 3, 1995.

Brief description of amendments: These amendments add the analytical method supplement entitled ''Fuel Rod Maximum Allowable Gas Pressure,'' CEN–372–P–A, dated May 1990, and its associated NRC Safety Evaluation, dated April 10, 1990, to the list of analytical methods in Technical Specification 6.9.1.10 used to determine the Palo Verde Nuclear Generating Station core operating limits.

Date of issuance: October 4, 1995.

Effective date: October 4, 1995, to be implemented prior to startup from RF06 for Units 1 and 2, and RF5 for Unit 3.

Amendment Nos.: Unit 1— Amendment No. 101; Unit 2— Amendment No. 89; Unit 3— Amendment No. 72.

Facility Operating License Nos. NPF–41, NPF–51, and NPF–74: The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: August 30, 1995 (60 FR 45173) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 4, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Phoenix Public Library, 1221 N. Central Avenue, Phoenix, Arizona 85004.

Baltimore Gas and Electric Company, Docket Nos. 50–317 and 50–318, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Calvert County, Maryland

Date of application for amendments: June 2, 1995.

Brief description of amendments: The amendments revise the tolerances for the pressurizer safety valve as-found acceptance criterion.

Date of issuance: September 26, 1995.

Effective date: As of the date of issuance to be implemented within 30 days.

Amendment Nos.: 206 and 184.

Facility Operating License Nos. DPR–53 and DPR–69: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: July 5, 1995 (60 FR 35060) The Commission's related evaluation of these amendments is contained in a Safety Evaluation dated September 26, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Calvert County Library, Prince Frederick, Maryland 20678.

Baltimore Gas and Electric Company, Docket Nos. 50–317 and 50–318, Calvert Cliffs Nuclear Power Plant, Unit No. 1, Calvert County, Maryland

Date of application for amendments: January 31, 1995.

Brief description of amendments: The amendments revise the Technical Specifications (TSs) to increase the amount of Trisodium Phosphate Dodecahydrate located in the containment sump baskets which is required to be verified by TS surveillance. The test requirements for verifying that the appropriate pH (acidity/alkalinity) would be maintained in the containment sump water following a design-basis accident are moved from the TSs to the TS Bases section; however, the requirement to perform the test remains in the TSs. The associated TS Bases sections are updated to reflect the changes.

Date of issuance: October 5, 1995.

Effective date: As of the date of issuance to be implemented within 30 days.

Amendment Nos.: 207 and 185.

Facility Operating License No. DPR–53 and DPR–69: Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: March 15, 1995 (60 FR 14016) The Commission's related evaluation of these amendments is contained in a Safety Evaluation dated October 5, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Calvert County Library, Prince Frederick, Maryland 20678

Commonwealth Edison Company, Docket Nos. 50–295 and 50–304, Zion Nuclear Power Station Units 1 and 2, Lake County, Illinois

Date of application for amendments: March 8, 1995, as supplemented on June 1, 1995.

Brief description of amendments: The amendments revise the secondary undervoltage setpoint.

Date of issuance: October 2, 1995.

Effective date: October 2, 1995

Amendment Nos.: 169 and 156.

Facility Operating License Nos. DPR–39 and DPR–48: The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: August 30, 1995 (60 FR 45178) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 2, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Waukegan Public Library, 128 N. County Street, Waukegan, Illinois 60085.

Consolidated Edison Company of New York, Docket No. 50–247, Indian Point Nuclear Generating Unit No. 2, Westchester County, New York

Date of application for amendment: February 18, 1994, as supplemented June 3, November 1, December 2, December 14, and December 16, 1994, and August 25, 1995.

Brief description of amendment: The amendment revises the surveillance intervals for the Boric Acid Tank Level, the Service Water Inlet Temperature Monitor Instrument, the Boric Acid Makeup Flow System, the Plant Noble Gas Activity Monitor, the Condenser Evacuation System Activity Monitor, the Low Turbine Auto Stop Oil Pressure Trip, the 6.9 kv Undervoltage Monitor, the Sampler Flow Rate Monitor, and the Refueling Water Storage Tank.

Date of issuance: October 12, 1995.

Effective date: As of the date of issuance to be implemented within 30 days.

Amendment No.: 184.

Facility Operating License No. DPR–26: Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: April 28, 1994 (59 FR 22003) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 12, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: White Plains Public Library, 100 Martine Avenue, White Plains, New York 10610.

Entergy Operations, Inc., Docket No. 50–368, Arkansas Nuclear One, Unit No. 2, Pope County, Arkansas

Date of application for amendment: March 17, 1995.

Brief description of amendment: The amendment revises requirements associated with channel functional tests of the core protection calculator following a high temperature alarm.

Date of issuance: October 11, 1995.

Effective date: October 11, 1995, to be implemented within 30 days.

Amendment No.: 168.

Facility Operating License No. NPF–6. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: August 2, 1995 (60 FR 39437) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 11, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Tomlinson Library, Arkansas Tech University, Russellville, AR 72801.

Entergy Operations, Inc., Docket No. 50–382, Waterford Steam Electric Station, Unit 3, St. Charles Parish, Louisiana

Date of amendment request: July 18, 1991, as supplemented by letters dated March 16, and December 2, 1994, and March 9, and August 30, 1995.

Brief description of amendment: The amendment changes the Appendix A TSs by subdividing TS 3/4.7.6, ''Control Room Air Conditioning System,'' into five separate TSs covering the following three distinct functions: control room emergency air filtration, control room air temperature, and control room isolation and pressurization. The amendment also changes the Bases sections of the TS to reflect the above changes.

Date of issuance: October 4, 1995.

Effective date: October 4, 1995.

Amendment No.: 115.

Facility Operating License No. NPF–38: Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: September 4, 1991 (56 FR 43808) and July 6, 1995 (60 FR 29875).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 4, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: University of New Orleans Library, Louisiana Collection, Lakefront, New Orleans, LA 70122.

GPU Nuclear Corporation, et al., Docket No. 50–289, Three Mile Island Nuclear Station, Unit No. 1, Dauphin County, Pennsylvania

Date of application for amendment: June 1, 1995, as supplemented August 23, 1995.

Brief description of amendment: The amendment changes the Technical Specifications to relocate the procedural details of the Radiological Effluent Technical Specifications to the Offsite Dose Calculation Manual. With these changes, the specifications related to RETS reporting requirements were simplified and changes to the definition of the ODCM were made to make the definition consistent with the amendment.

Date of Issuance: October 2, 1995.

Effective date: As of the date of issuance to be implemented within 120 days.

Amendment No.: 197.

Facility Operating License No. DPR–50. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: July 5, 1995 (60 FR 35078) The August 23, 1995, letter provided supplemental information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated October 2, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Law/Government Publications Section, State Library of Pennsylvania, (Regional Depository) Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, PA 17105.

GPU Nuclear Corporation, et al., Docket No. 50–289, Three Mile Island Nuclear Station, Unit No. 1, Dauphin County, Pennsylvania

Date of application for amendment: January 16, 1995, as supplemented June 22 and September 20, 1995.

Brief description of amendment: The amendment revises the Technical Specifications for TMI–1 to incorporate seven improvements from the Revised Standard Technical Specifications for Babcock & Wilcox Nuclear Power Plants (NUREG–1430). The amendment also changes the Bases incorporating the results of analyses to support allowance for drift of the Pressurizer Code Safety Valve setpoint. The remaining portion of the request relating to revisions to Control Room Emergency Ventilation system are being reviewed separately.

Date of Issuance: October 10, 1995.

Effective date: October 10, 1995.

Amendment No.: 198.

Facility Operating License No. DPR–50. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: March 15, 1995 (60 FR 14021). The June 22 and September 20, 1995, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated October 10, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Law/Government Publications Section, State Library of Pennsylvania, (Regional Depository) Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, PA 17105.

Indiana Michigan Power Company, Docket Nos. 50–315 and 50–316, Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2, Berrien County, Michigan

Date of application for amendments: November 12, 1993, as supplemented November 18, 1994, May 30, 1995, and August 8, 1995.

Brief description of amendments: The amendments delete from the Technical Specifications the sections and tables entitled ''Component Cyclic or Transient Limits'' and relocate the information to the Updated Final Safety Analysis Report.

Date of issuance: September 28, 1995.

Effective date: September 28, 1995, with full implementation within 45 days.

Amendment Nos.: 201 and 186.

Facility Operating License Nos. DPR–58 and DPR–74. Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: December 22, 1993 (58 FR 67849). The November 18, 1994, May 30, 1995, and August 8, 1995, supplements provided clarifying information and corrections to additional pages which referenced the table to be deleted. This information was within the scope of the original application and did not change the staff's initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 28, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Maud Preston Palenske Memorial Library, 500 Market Street, St. Joseph, Michigan 49085.

Indiana Michigan Power Company, Docket Nos. 50–315 and 50–316, Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2, Berrien County, Michigan

Date of application for amendments: May 26, 1995.

Brief description of amendments: The amendments modify Technical Specification Sections 3/4.3.1 and 3/4.3.2 and their accompanying Bases, to relocate the tables of response time limits for the reactor trip system and engineered safety feature acutation system instrumentation to the Updated Final Safety Analysis Report.

Date of issuance: October 10, 1995.

Effective date: October 10, 1995.

Amendment Nos.: 202 and 187.

Facility Operating License Nos. DPR–58 and DPR–74. Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: July 5, 1995 (60 FR 35082) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 10, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Maud Preston Palenske Memorial Library, 500 Market Street, St. Joseph, Michigan 49085.

Maine Yankee Atomic Power Company, Docket No. 50–309, Maine Yankee Atomic Power Station, Lincoln County, Maine

Date of application for amendment: May 5, 1995.

Brief description of amendment: The amendment revises the surveillance frequency of radiation area, and effluent and process monitors from monthly to quarterly; and the required frequency for minimum exercise of control element assemblies also from monthly to quarterly.

Date of issuance: October 2, 1995.

Effective date: As of the date of issuance, to be implemented within 30 days.

Amendment No.: 153.

Facility Operating License No. DPR–36: Amendment revised the Technical Specifications and/or License.

Date of initial notice in Federal Register: August 30, 1995 (60 FR 45179). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 2, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Wiscasset Public Library, High Street, P.O. Box 367, Wiscasset, ME 04578.

Northern States Power Company, Docket No. 50–263, Monticello Nuclear Generating Plant, Wright County, Minnesota

Date of application for amendment: June 8, 1994, as superseded by letter dated April 20, 1995, and supplemented by letter dated August 18, 1995.

Brief description of amendment: The amendment revises Sections 3.7/4.7, which pertain to the standby gas treatment system (SGTS) and secondary containment. The amendment revises the surveillance requirements for both SGTS and the secondary containment and revises the performance requirements for the SGTS filters and process stream electric heaters.

Date of issuance: October 2, 1995.

Effective date: October 2, 1995.

Amendment No.: 94.

Facility Operating License No. DPR–22. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: July 20, 1994 (59 FR 37075). The April 20 and August 18, 1995, submittals provided clarifying information within the scope of the original submittal and did not change the staff's initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 2, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Minneapolis Public Library, Technology and Science Department, 300 Nicollet Mall, Minneapolis, Minnesota 55401

Northern States Power Company, Docket Nos. 50–282 and 50–306, Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2, Goodhue County, Minnesota.

Date of application for amendments: July 11, 1994, as supplemented April 18, 1995 (supersedes the February 10, 1993, application).

Brief description of amendments: The amendments change license condition 2.C.(4) of each license to conform to the standard fire protection license condition as stated in Generic Letter 86–10. In addition, the amendments delete fire protection program elements from the Technical Specifications and incorporate, by reference, the NRC-approved Fire Protection Program and major commitments, including the fire hazards analysis, into the Updated Safety Analysis Report.

Date of issuance: October 6, 1995.

Effective date: October 6, 1995, with full implementation within 30 days.

Amendment Nos.: 120 and 113.

Facility Operating License Nos. DPR–42 and DPR–60. Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: December 21, 1994 (59 FR 65818). The April 18, 1995, letter provided clarifying information within the scope of the original submittal and did not change the staff's initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 6, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Minneapolis Public Library, Technology and Science Department, 300 Nicollet Mall, Minneapolis, Minnesota 55401.

Sacramento Municipal Utility District, Docket No. 50–312, Rancho Seco Nuclear Generating Station, Sacramento County, California

Date of application for amendment: June 20, 1995.

Brief description of amendment: This amendment modifies the technical specifications on spent fuel storage building load handling limits to allow the placement of the top shield plug on a dry shielded canister containing spent fuel which is being prepared for transfer to the Rancho Seco Independent Spent Fuel Storage Installation.

Date of issuance: October 5, 1995.

Effective date: October 5, 1995.

Amendment No.: 123.

Facility Operating License No. NPF–1: The amendment revised the Technical Specifications.

Date of initial notice in Federal Register: August 30, 1995 (60 FR 45184). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 5, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Central Library, Government Documents, 828 I Street, Sacramento, California 95814.

South Carolina Electric & Gas Company, South Carolina Public Service Authority, Docket No. 50–395, Virgil C. Summer Nuclear Station, Unit No. 1, Fairfield County, South Carolina

Date of application for amendment: June 30, 1995, as supplemented on August 11, 1995.

Brief description of amendment: The amendment revises the Technical Specifications (TS) for the pressurizer power operated relief valves to follow the NRC's guidance of Generic Letter 90–06 (Generic Issue 70), and the improved Westinghouse Standard TS (NUREG–1431, Rev. 1).

Date of issuance: September 18, 1995.

Effective date: September 18, 1995.

Amendment No.: 129.

Facility Operating License No. NPF–12. Amendment revises the TS.

Date of initial notice in Federal Register: Auust 16, 1995 (60 FR 42608).

The August 11, 1995, supplemental letter corrected an error in the original submittal and did not change the initial proposed no significant hazards consideration. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 18, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Fairfield County Library, 300 Washington Street, Winnsboro, SC 29180.

Southern Nuclear Operating Company, Inc., Docket Nos. 50–348 and 50–364, Joseph M. Farley Nuclear Plant, Units 1 and 2, Houston County, Alabama

Date of amendments request: August 17, 1994, as supplemented by letters dated June 15 and August 11, 1995.

Brief Description of amendments: The amendments eliminate periodic pressure sensor response time testing surveillance requirements for specific Reactor Trip System and Engineered Safety Feature Actuation System instrumentation specified in Technical Specification Sections 4.3.1.3 and 4.3.2.3.

Date of issuance: September 28, 1995.

Effective date: As of the date of issuance to be implemented within 30 days from the date of issuance.

Amendment Nos.: 116 and 108.

Facility Operating License Nos. NPF–2 and NPF–8. Amendments revise the Technical Specifications.

Date of initial notice in Federal Register: September 28, 1994 (59 FR 49434) The June 15 and August 11, 1995, letters provided clarifying information that did not change the scope of the August 17, 1994, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 28, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Houston-Love Memorial Library, 212 W. Burdeshaw Street, Post Office Box 1369, Dothan, Alabama 36302.

Southern Nuclear Operating Company, Inc., Docket No. 50–348, Joseph M. Farley Nuclear Plant, Unit 1, Houston County, Alabama

Date of amendment request: December 7, 1994, as supplemented by letter dated May 31, 1995.

Brief Description of amendment: The amendment revised Farley Unit 1 Technical Specifications 4.4.6.2, 4.4.6.4, 4.4.6.5, 3.4.7.2, and 3.4.9 for Cycle 14 operation to permit the use of steam generator tube repair criteria for defects confined within the thickness of the tube support plate.

Date of issuance: September 28, 1995.

Effective date: As of the date of issuance to be implemented prior to the start of Unit 1, Cycle 14 operation.

Amendment No.: 117.

Facility Operating License No. NPF–2: Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: February 15, 1995 (60 FR 8754) The May 31, 1995, letter provided clarifying information that did not change the scope of the December 7, 1994, application and the initial proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 28, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Houston-Love Memorial Library, 212 W. Burdeshaw Street, Post Office Box 1369, Dothan, Alabama 36302.

Tennessee Valley Authority, Docket Nos. 50–327 and 50–328, Sequoyah Nuclear Plant, Units 1 and 2, Hamilton County, Tennessee

Date of application for amendments: August 7, 1995 (TS 95–18).

Brief description of amendments: The amendments revise the titles of various administrative positions found in Section 6.0 of the Technical Specifications.

Date of issuance: October 2, 1995.

Effective date: October 2, 1995,

Amendment Nos.: 212 and 202.

Facility Operating License Nos. DPR–77 and DPR–79: Amendments revise the technical specifications.

Date of initial notice in Federal Register: August 30, 1995 (60 FR 45186) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 2, 1995.

No significant hazards consideration comments received: None.

Local Public Document Room location: Chattanooga-Hamilton County Library, 1101 Broad Street, Chattanooga, Tennessee 37402.

Tennessee Valley Authority, Docket Nos. 50–327 and 50–328, Sequoyah Nuclear Plant, Units 1 and 2, Hamilton County, Tennessee

Date of application for amendments: August 7, 1995 (TS 95–12).

Brief description of amendments: The amendments correct various editorial errors in the text of the technical specifications and remove provisions that have expired or are no longer applicable.

Date of issuance: October 4, 1995.

Effective date: October 4, 1995.

Amendment Nos.: 213 and 203.

Facility Operating License Nos. DPR–77 and DPR–79: Amendments revise the technical specifications.

Date of initial notice in Federal Register: August 30, 1995 (60 FR 45185) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 4, 1995.

No significant hazards consideration comments received: None.

Local Public Document Room location: Chattanooga-Hamilton County Library, 1101 Broad Street, Chattanooga, Tennessee 37402.

Tennessee Valley Authority, Docket No. 50–327, Sequoyah Nuclear Plant, Unit 1, Hamilton County, Tennessee

Date of application for amendment: July 19, 1995, superseded September 7, 1995 and supplemented September 15 and 26, 1995 (TS 95–15).

Brief description of amendment: The amendment revises the TS surveillance requirements and bases to incorporate alternate S/G tube plugging criteria at tube support plate (TSP) intersections. The approach taken is similar to guidance given in Generic Letter (GL) 95–05, ''Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking.''

Date of issuance: October 11, 1995.

Effective date: October 11, 1995.

Amendment No.: 214.

Facility Operating License Nos. DPR–77: Amendment revises the technical specifications.

Date of initial notice in Federal Register: August 1, 1995 (60 FR 39189) The letters dated September 7, 15 and 26, 1995 provided information that did not change the initial proposed no significant hazards consideration. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 11, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Chattanooga-Hamilton County Library, 1101 Broad Street, Chattanooga, Tennessee 37402.

Virginia Electric and Power Company, et al., Docket Nos. 50–338 and 50–339, North Anna Power Station, Units No. 1 and No. 2, Louisa County, Virginia

Date of application for amendments: March 30, 1995, as supplemented August 24, 1995

Brief description of amendments: The amendments revise the North Anna 1 and 2 Technical Specifications to allow one of the two service water loops to be isolated from the component cooling water head exchangers during power operations in order to refurbish the isolated service water headers.

Date of issuance: October 11, 1995.

Effective date: October 11, 1995.

Amendment Nos.: 194 and 175.

Facility Operating License Nos. NPF–4 and NPF–7. Amendments revised the Technical Specifications.

Date of initial notice in Federal Register*:* May 10, 1995 (60 FR 24923). The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated October 11, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: The Alderman Library, Special Collections Department, University of Virginia, Charlottesville, Virginia 22903–2498.

Washington Public Power Supply System, Docket No. 50–397, Nuclear Project No. 2, Benton County, Washington

Date of application for amendment: December 6, 1993.

Brief description of amendment: The amendment changes the surveillance requirements in Technical Specification 4.6.6.1.b.3 to provide more appropriate acceptance criteria for demonstrating operability of the primary containment hydrogen recombiner systems.

Date of issuance: October 5, 1995.

Effective date: October 5, 1995, to be implemented within 30 days of issuance.

Amendment No.: 142.

Facility Operating License No. NPF–21: The amendment revised the Technical Specifications.

Date of initial notice in Federal Register*:* July 6, 1994 (59 FR 34670). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 5, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room location: Richland Public Library, 955 Northgate Street, Richland, Washington 99352.

Wolf Creek Nuclear Operating Corporation, Docket No. 50–482, Wolf Creek Generating Station, Coffey County, Kansas

Date of amendment request: May 24, 1994, as supplemented by letter dated April 6, 1995.

Brief description of amendment: This amendment revises the technical specifications (TS) to implement the NRC's revised 10 CFR 50.36 on technical specification improvements for nuclear power reactors. Specifications that do not meet any of the four criteria or regulatory requirements related to inclusion in the TS are relocated to Chapter 16 of the Updated Safety Analysis Report.

Date of issuance: October 2, 1995.

Effective date: October 2, 1995, to be implemented within 120 days from the date of issuance.

Amendment No.: 89.

Facility Operating License No. NPF–42. The amendment revised the Technical Specifications.

Date of initial notice in Federal Register*:* July 6, 1994 (59 FR 34671). The April 6, 1995, supplemental letter provided additional clarifying information and did not change the initial no significant hazards consideration determination.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 2, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room locations: Emporia State University, William Allen White Library, 1200 Commercial Street, Emporia, Kansas 66801 and Washburn University School of Law Library, Topeka, Kansas 66621.

Wolf Creek Nuclear Operating Corporation, Docket No. 50–482, Wolf Creek Generating Station, Coffey County, Kansas

Date of amendment request: July 25, 1995.

Brief description of amendment: The amendment deletes a clause from Section 4.0.5a, ''Surveillance Requirements for Inservice Inspection and Testing Program.'' This clause required prior NRC approval before implementation of a relief request upon finding an ASME Code requirement impractical because of prohibitive dose rates or limitations in the design, construction, or system configuration.

Date of issuance: October 4, 1995.

Effective date: October 4, 1995, to be implemented within 30 days of issuance.

Amendment No.: 90.

Facility Operating License No. NPF–42. The amendment revised the Technical Specifications.

Date of initial notice in Federal Register*:* August 30, 1995 (60 FR 45191). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated October 4, 1995.

No significant hazards consideration comments received: No.

Local Public Document Room locations: Emporia State University, William Allen White Library, 1200 Commercial Street, Emporia, Kansas 66801 and Washburn University School of Law Library, Topeka, Kansas 66621

Notice of Issuance of Amendments to Facility Operating Licenses and Final Determination of No Significant Hazards Consideration and Opportunity for a Hearing (Exigent Public Announcement or Emergency Circumstances)

During the period since publication of the last biweekly notice, the Commission has issued the following amendments. The Commission has determined for each of these amendments that the application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Because of exigent or emergency circumstances associated with the date the amendment was needed, there was not time for the Commission to publish, for public comment before issuance, its usual 30-day Notice of Consideration of Issuance of Amendment, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing.

For exigent circumstances, the Commission has either issued a Federal Register notice providing opportunity for public comment or has used local media to provide notice to the public in the area surrounding a licensee's facility of the licensee's application and of the Commission's proposed determination of no significant hazards consideration. The Commission has provided a reasonable opportunity for the public to comment, using its best efforts to make available to the public means of communication for the public to respond quickly, and in the case of telephone comments, the comments have been recorded or transcribed as appropriate and the licensee has been informed of the public comments.

In circumstances where failure to act in a timely way would have resulted, for example, in derating or shutdown of a nuclear power plant or in prevention of either resumption of operation or of increase in power output up to the plant's licensed power level, the Commission may not have had an opportunity to provide for public comment on its no significant hazards consideration determination. In such case, the license amendment has been issued without opportunity for comment. If there has been some time for public comment but less than 30 days, the Commission may provide an opportunity for public comment. If comments have been requested, it is so stated. In either event, the State has been consulted by telephone whenever possible.

Under its regulations, the Commission may issue and make an amendment immediately effective, notwithstanding the pendency before it of a request for a hearing from any person, in advance of the holding and completion of any required hearing, where it has determined that no significant hazards consideration is involved.

The Commission has applied the standards of 10 CFR 50.92 and has made a final determination that the amendment involves no significant hazards consideration. The basis for this determination is contained in the documents related to this action. Accordingly, the amendments have been issued and made effective as indicated.

Unless otherwise indicated, the Commission has determined that these amendments satisfy the criteria for categorical exclusion in accordance with 10 CFR 51.22. Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared for these amendments. If the Commission has prepared an environmental assessment under the special circumstances provision in 10 CFR 51.12(b) and has made a determination based on that assessment, it is so indicated.

For further details with respect to the action see (1) The application for amendment, (2) the amendment to Facility Operating License, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment, as indicated. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room for the particular facility involved.

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By November 24, 1995, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

As required by 10 CFR 2.714, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following factors: (1) The nature of the petitioner's right under the Act to be made a party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to 15 days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

Not later than 15 days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner shall provide a brief explanation of the bases of the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner intends to rely to establish those facts or expert opinion. Petitioner must provide sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses. Since the Commission has made a final determination that the amendment involves no significant hazards consideration, if a hearing is requested, it will not stay the effectiveness of the amendment. Any hearing held would take place while the amendment is in effect.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, by the above date. Where petitions are filed during the last 10 days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1–(800) 248–5100 (in Missouri 1–(800) 342–6700). The Western Union operator should be given Datagram Identification Number N1023 and the following message addressed to *(Project Director)*: petitioner's name and telephone number, date petition was mailed, plant name, and publication date and page number of this Federal Register notice. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to the attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for a hearing will not be entertained absent a determination by the Commission, the presiding officer or the Atomic Safety and Licensing Board that the petition and/or request should be granted based upon a balancing of the factors specified in 10 CFR 2.714(a)(1)(i)–(v) and 2.714(d).

Detroit Edison Company, Docket No. 50–341, Fermi–2, Monroe County, Michigan

Date of application for amendment: October 2, 1995.

Brief description of amendment: The amendment revises the Technical Specifications to allow deferral until the next plant outage of certain portions of logic system functional surveillance testing for the diesel generator 480-volt load sequencer and output breaker reclosure logic circuitry.

Date of issuance: October 13, 1995.

Effective date: October 13, 1995, with full implementation within 45 days.

Amendment No.: 105.

Facility Operating License No. NPF–43: Amendment revises the Technical Specifications.

Public comments requested as to proposed no significant hazards consideration: No.

The Commission's related evaluation of the amendment, finding of emergency circumstances, and final determination of no significant hazards consideration are contained in a Safety Evaluation dated October 13, 1995.

Local Public Document Room location: Monroe County Library System, 3700 South Custer Road, Monroe, Michigan 48161.

Attorney for licensee: John Flynn, Esq., Detroit Edison Company, 2000 Second Avenue, Detroit, Michigan 48226.

NRC Project Director: Brian E. Holian, Acting.

Pacific Gas and Electric Company, Docket Nos. 50–275 and 50–323, Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2, San Luis Obispo County, California

Date of application for amendments: September 30, 1995.

Brief description of amendments: The amendments increase the setpoint tolerance of the main steam safety valves (MSSVs) from plus or minus 1 percent to plus or minus 3 percent, with the exception that the lowest set MSSVs would have a tolerance of −2 percent/+3 percent.

Date of issuance: October 1, 1995.

Effective date: October 1, 1995.

Amendment Nos.: Unit 1—Amendment No. 108; Unit 2—Amendment No. 107.

Facility Operating License Nos. DPR–80 and DPR–82: The amendments revised the Technical Specifications.

Public comments requested as to proposed no significant hazards consideration: No.

The Commission's related evaluation of the amendment, finding of emergency circumstances, and final determination of no significant hazards consideration are contained in a Safety Evaluation dated October 1, 1995.

Local Public Document Room location: California Polytechnic State University, Robert E. Kennedy Library, Government Documents and Maps Department, San Luis Obispo, California 93407.

Attorney for licensee: Christopher J. Warner, Esq., Pacific Gas and Electric Company, P.O. Box 7442, San Francisco, California 94120.

NRC Project Director: William H. Bateman.

Dated at Rockville, Maryland, this 18th day of October 1995.

For the Nuclear Regulatory Commission.

Elinor G. Adensam,

Deputy Director, Division of Reactor Projects—III/IV, Office of Nuclear Reactor Regulation.

[FR Doc. 95–26275 Filed 10–24–95; 8:45 am]
BILLING CODE 7590–01–P

[Docket No. 27–48]

Consideration of an Application for Renewal of a License to Dispose of Low-Level Radioactive Waste Containing Special Nuclear Material by American Ecology Corporation and Opportunity for a Hearing

AGENCY: Nuclear Regulatory Commission.

ACTION: Consideration of an application for renewal of a license to dispose of low-level radioactive waste (LLW) containing special nuclear material (SNM) by American Ecology Corporation and opportunity for a hearing.

SUMMARY: The Nuclear Regulatory Commission is considering the renewal of License No. 16–19204–01. This license is issued to American Ecology Corporation for the disposal of wastes containing SNM in the low-level radioactive waste disposal facility, located on the Hanford Reservation near Richland, WA. The license is currently under timely renewal. NRC licenses this facility under 10 CFR Part 70. The license renewal application was tendered on October 28, 1993. NRC has delayed review of the application pending allocation of sufficient resources to conduct the review.

FOR FURTHER INFORMATION CONTACT: Robert A. Nelson, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Telephone: (301) 415–7298, Fax.: (301) 415–5397.

Background

The LLW disposal facility located on the Hanford Reservation in Benton County, Washington, is licensed by the State of Washington for disposal of source and byproduct material. The NRC license allows the disposal of SNM, and acknowledges the State regulated activities constitute the major site activities. As a result, NRC relies extensively on the State's regulatory program to evaluate the facility and licensee's capability to demonstrate reasonable assurance that the disposal of LLW can be accomplished safely. To this end, NRC coordinates review and assessment of the licensee with the State of Washington, Department of Health. To avoid duplicative effort, NRC has identified areas in which it relies primarily on the State regulatory program. Areas distinct to SNM regulation are directly evaluated by NRC. Under the NRC license several State identified license conditions are referenced, this ensures that NRC is aware of significant licensee activities requiring State regulatory action. Additionally, NRC incorporates conditions in the SNM license which provide NRC the lattitude to enforce the Agreement State license conditions, if NRC determines that the Agreement State is not enforcing the license conditions. Finally, the NRC license does not abrogate or diminish the authority of the State of Washington, under its Agreement under section 274b